Exelon Generation Company, LLC
Dresden Nuclear Power Station
6500 North Dresden Road
Morris, IL 60450-9765

www.exeloncorp.com

Exelon Nuclear

10 CFR 50.73

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Dresden Nuclear Power Station, Unit 3
Facility Operating License No. DPR-25
NRC Docket No. 50-249

Subject: Licensee Event Report 2001-003-00, "Reactor Scram due to Increasing Drywell Pressure"

Enclosed is Licensee Event Report 2001-003-00, "Reactor Scram due to Increasing Drywell Pressure," for the Dresden Nuclear Power Station (DNPS). This condition is being reported pursuant to 10 CFR 50.73 (a)(2)(iv)(B), which requires the reporting of any event or condition that resulted in a manual or automatic actuation of the Reactor Protection System (RPS) including reactor scram or reactor trip.

The following actions were taken:

Evaluated and implemented the operation of two Reactor Building Closed Cooling Water (RBCCW) pumps and two RBCCW heat exchangers per unit.

Disassembled and repaired the 3B RBCCW Temperature Control Valve (TCV).

This correspondence contains the following new commitments:

Disassemble and install the correct retaining pin properly in the remaining RBCCW TCVs for Unit 2 and 3. Verify that the TCV stem is properly torqued to its disc.

Develop a plan for long term operation of the two RBCCW pumps and heat exchangers.

Any other actions described in the submittal represent intended or planned actions by DNPS. They are described for the NRC's information and are not regulatory commitments.

IE22

If you have any questions, please contact Mr. Dale F. Ambler, Dresden Regulatory Assurance Manager at (815) 416-2800.

Respectfully,

R Fisher for PDS

Preston Swafford
Site Vice President
Dresden Nuclear Power Station

Enclosure

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the information and Records Management Branch (t-6 f33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office Of Management And Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Reactor SCRAM due to Increasing Drywell Pressure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	N/A
07	05	2001	2001	003	00	09	04	2001	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(l)		20.2203(a)(3)(l)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(l))		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Timothy P. Heisterman, Regulatory Assurance	(815) 416 3324

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	KG	TCV	C635	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i. e., approximately 15 single-spaced typewritten lines) (16)

On July 5, 2001, at 1006 hours, a manual Scram was initiated on Dresden Nuclear Power Station (DNPS) Unit 3 due to increasing drywell pressure. Drywell pressure increased because of rising containment temperature due to a disc to stem separation of the Reactor Building Closed Cooling Water (RBCCW) Temperature Control Valve (TCV) that supplies cooling water to the RBCCW heat exchangers. One of the RBCCW loads is the drywell coolers. After the manual scram was initiated, drywell pressure continued to increase to a value above the Emergency Core Cooling System (ECCS) initiation setpoint. ECCS systems automatically initiated at 1.7 psig drywell pressure. The High Pressure Coolant Injection (HPCI) was placed in pull-to-lock to prevent a cold water injection. A Group 2 and Group 3 isolation also occurred during this event and all isolation valves successfully closed or isolated. All ECCS initiated and performed as designed.

A General Station Emergency Plan (GSEP) Alert was conservatively declared based on the assumption that the drywell pressure increased due to Reactor Coolant System (RCS) leakage. Subsequently, containment sampling and sump data determined that drywell atmospheric activity levels had remained normal, and no abnormal sump inputs had occurred (no indication of RCS leakage). The GSEP Alert was terminated on July 5, 2001 at 1602 hours.

The root cause of this event was the inappropriate torqueing of the valve stem into the valve disc by the manufacturer of the plug assembly. The cause of the manufacturer's failure to properly assemble the valve is unknown. The contributing causes of this event are poor communication of a similar previous event at Quad Cities, improper pin material being supplied by the vendor and Dresden's dependency on a single RBCCW train.

The safety significance of this event has been determined to be minimal.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Dresden Nuclear Power Station, Unit 3	05000249	2001	003	00	2 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

PLANT AND SYSTEM IDENTIFICATION:

General Electric – Boiling Water Reactor – 2527 MWt rated core thermal power
Energy Industry Identification System (EIIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Reactor SCRAM due to Increasing Drywell Pressure

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3	Event Date: 07-05-2001	Event Time: 1006
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 100%

Reactor Coolant System Pressure: 1000 psig

B. DESCRIPTION OF EVENT:

This condition is being reported pursuant to 10 CFR 50.73 (a)(2)(iv)(B), which requires the reporting of any event or condition that resulted in a manual or automatic actuation of the Reactor Protection System (RPS) [JC] including reactor scram or reactor trip.

On July 5, 2001 at 1000 hours, the Unit 3 Nuclear Station Operator (NSO) observed drywell to torus differential pressure was slightly elevated during routine panel monitoring. Both indications, drywell pressure and drywell to torus differential pressure, were indicating approximately 1.3 psig on Unit 3. Because atmospheric conditions could cause this response, the Unit 3 NSO communicated with the Unit 2 NSO for comparison of drywell differential pressure. The Unit 2 drywell pressure was approximately 1.14 psig and stable. At this time, an additional NSO was requested to assist in the determination of the indicated elevated drywell pressure.

As pressure continued to increase, the NSOs consulted the Unit Supervisor. The Unit Supervisor advised the NSO to manually Scram the reactor when drywell pressure reached 1.5 psig. At this time, there were no other abnormal alarm indications.

Drywell temperature, and therefore drywell pressure continued to increase, and at 1006 hours a manual Scram was inserted at 1.49 psig. Drywell pressure continued to slowly increase. At 1009 hours the pressure reached 1.71 psig, at which time Emergency Core Cooling System (ECCS) equipment initiated on drywell high pressure (Unit 2/3 and Unit 3 diesel generators [EK] ran unloaded, Core Spray [BM] and LPCI [BO] initiated and operated in recirculation mode). High Pressure Coolant Injection (HPCI) [BJ] was placed in pull-to-lock to prevent a cold water injection. All ECCS initiated and operated as required.

At 1019 hours, a GSEP Alert level was conservatively declared based on the assumption that the drywell pressure increased due to RCS leakage. At 1036 hours a pressure of 2.0 psig was reached and peaked at 2.3 psig. Based on observations by Operations during the event, the potential for failure of the 3B RBCCW [CC] TCV existed. At 1113 hours, the 3A RBCCW pump was started and aligned to the 3A RBCCW heat exchanger. At 1128 hours drywell pressure had decreased to less than 2.0 psig (one and a half-hours after the initiation of the event).

At 1435 hours, drywell gaseous atmospheric sample indicated normal airborne activity levels, which was indicative of no abnormal RCS leakage. At 1500 hours, Operations reported drywell sumps indicate normal leakage (indicative of no RCS leakage). At 1530 hours a maintenance crew performed testing that indicated that the 3B RBCCW Heat Exchanger Service Water Outlet TCV (throttles service water) appeared to have some kind of disc separation/damage, interrupting flow. The Alert was terminated at 1602 hours, on July 5, 2001.

C. CAUSE OF EVENT:

Subsequent investigation revealed that the Unit 3B TCV had failed due to inappropriate torqueing of the valve stem into the valve disc. This inappropriate action led to the disc dropping into the seat of the valve and thus obstructing the flow path of cooling water to the RBCCW heat exchanger.

Upon disassembly of the valve, it was noted that the holes for the roll pin lined up when the stem was threaded hand-tight into the disc. When the stem was properly torqued into the disc, the holes did not align. This indicates that the stem was not torqued into the disc, producing additional stresses on the pin due to flow through the system (i.e. vibration). Because stress is one component of stress corrosion cracking, this additional stress is a key component of the failure. If the disc had been properly torqued, there would be no additional stresses on the pin and this event would not have occurred. The manufacturer's valve assembly procedure requires that the stem be torqued into the plug and a solid 300 series Stainless Steel pin to be used. This has been determined to be the root cause of this event. (NRC Cause Code B)

A contributing cause to the event was that prior to 1999, the RBCCW system was operated in a parallel configuration, with a two-pump/heat exchanger combination. In 1999, after RBCCW system was balanced, it was determined that only a one-pump/heat exchanger was required to be in operation.

D. SAFETY ANALYSIS

Although the Unit was manually scrammed as a result of increasing pressure in the Drywell, this event was of minimal safety significance. All rods fully inserted during the manual scram, all Group 2 and Group 3 isolation valves successfully closed or isolated, and all ECCS responded satisfactorily. Reactor level was maintained using normal Feedwater control and Main Turbine bypass valves and the Main Condenser was used to remove decay heat.

A shutdown risk assessment was performed that showed that the overall window and all individual windows remained GREEN for the duration of the outage. There was no impact on the non-outage unit and the Technical Specifications were met. Therefore, the safety significance of this event is minimal.

E. CORRECTIVE ACTIONS:

Disassembled and repaired the 3B RBCCW TCV. (Complete)

Evaluated and implemented the operation of two RBCCW pumps and two heat exchangers per unit. (Complete)

Develop a plan for long term operation of the two RBCCW pumps and heat exchangers. (ATI 56390-10)

Disassemble and install the correct retaining pin properly in the remaining RBCCW TCVs. Verify that the TCV stem is properly torqued to its disc. (ATI 56390-11)

F. PREVIOUS OCCURRENCES:

There was a previous similar occurrence at Quad Cities. The failure of their RBCCW TCV did not result in a unit scram since two RBCCW heat exchangers per unit were operating at the time. A Nuclear Operations Notification (NON) was not generated as a result of the Quad Cities event. An OPEX search was performed and no previous similar events have been reported. A similar search was performed on Dresden's Condition Reporting database, and no previous events have been reported.

G. COMPONENT FAILURE DATA:

Copes-Vulcan Temperature Control Valve Class 125, D style control valve with 600-160L direct acting actuator, with anti-cavitation trim.